Nine Mile Point, Unit 1 Loss of Shutdown Cooling Event:

Citations Regarding Loss of Barriers to Radiation Release During the Event

(Cover letter, 2nd Par.)

The enclosed best estimate analysis, given that the event occurred early in the outage while no automatic injection systems were available and the primary containment was not functional, preliminarily determined that the increase in core damage frequency (CDF) and a large early release frequency (LERF) of radioactive material were in the range of one in one million years and one in ten million years, respectively. The analysis estimated the chance that operators could fail to restore SDC and then also fail to add water to the reactor coolant system to account for the amount that would have boiled, prior to core damage. The most influential input for both CDF and LERF was use of current SDP risk analysis guidance which limits the combination of human error probabilities to a chance of one in one million. Relating to LERF, assumptions concerning the point at which the evacuation of the population close to the plant would be initiated relative to the time of core damage were most influential.

(Page A-2)

3.0 Plant Conditions Prior to the Event

Plant equipment and conditions were as follows:

- Shutdown cooling (SDC) pump 12 and associated support systems were inservice cooling the core. This was the "protected" train
- SDC pumps 11 and 13 were non-functional with their breakers racked out. However, the pumps were available with manual actions outside the control room
- Reactor water cleanup was inservice letting down at approximately 40 gpm
- Condensate system was inservice making up to the reactor at approximately 40 gpm
- Both trains (all four pumps) of core spray were non-functional, i.e., the pumps would not auto start on low reactor water level. However, the pumps could be started with manual actions outside the control room
- Both control rod drive pumps were non-functional but could be manually started with operator actions outside the control room
- Fire water system was available
- Containment spray (including torus cooling) was non-functional but could be started with actions outside the control room
- Containment Spray Raw water system was available
- Reactor water level was at the flange
- Reactor head vent piping was removed, thus ensuring that the reactor coolant system would not pressurize on a loss of shutdown cooling
- Estimated time to boil prior to the event was calculated to be less than 2 hours. The heat up from the actual event, 115 F to 145 F in approximately 30 minutes, indicates an actual time to

boil of about 110 minutes Estimated time to core uncovery (the surrogate for core damage) was about 9 hours.

- Primary containment was open and not restorable

(Page A-6)
The LERF event tree is in FigureC-1 of Appendix C. The review of the LERF event indicates a negative response for all of the event tree tops with one exception. A detailed review of the LERF event tree and each of the event tree tops is also presented in Appendix C. The questions that are answered in the negative include:

- Greater than 8 days after shutdown: It was only 1.5 days after shutdown
- Containment Isolated and not bypassed: Primary containment was open and not closable
- Containment inerted: No
- Water on drywell floor: No (but not relevant – see detailed discussion in Appendix C)
- Core damage arrested without vessel breach: Vessel head vent piping was removed therefore, by definition the vessel was breached
- No containment failure at vessel breach: Containment was open as the containment (drywell) head was removed and the head vent piping removed
- No venting after vessel breach: Containment was vented as the containment (drywell) head was removed and the head vent piping removed

The two questions that determine the LERF conclusion relate to the possibility of evacuating the near-in population before core damage and containment failure. As the containment is failed because of the containment head removal the question that needs to be resolved is the probability of evacuating the near-in population. The licensee determined that the emergency plan requires the declaration of a general emergency upon core uncovery, and that under the existing weather conditions at the time of the event it would take approximately two hours to evacuate the near-in population. The analyst has determined that the time to boil off the water to the bottom of active fuel is approximately two hours. Therefore, the time from core uncovery to core damage is less than two hours and therefore, the entire near-in population, as currently understood, cannot be evacuated.

(Pages A-10 to A-11)
In addition to the quantitative assessm ent the licensee performed a qualitative assessment of the event and their response capability. The NRC analyst performed a cursory deterministic analysis against Regulatory Guide (RG) 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 2. In Section 2.1.1 this RG supplies a concise definition of defense-in-depth. Each of the defense-in-depth attributes is discussed below.

- A reasonable balance is preserved among prevention of core damage, containment failure and consequence mitigation: As the event has already initiated and the containment is open (a requirement in BWR Mark I containments to refuel) all of the reliance is on mitigation.

- Over-reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided: The only way to mitigate this event was a reliance on programmatic activities (i.e., operator action).
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges, and uncertainties: System redundancy and diversity were maintained. However, because of the loss of DC bus 12, an entire train of safety related equipment was lost to the operators initially. This lost equipment was recoverable but it took the operators over an hour to do so. Based on operating experience, losses of shutdown cooling occur about once per shutdown year. Therefore, these are not rare events and should be anticipated.
- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed: By removing all automatic actions a common cause failure – failure of the operators – was introduced.
- Independence of barriers is not degraded: Primary containment was open and the RCS was breached. Thus two out of the three barriers were not available prior to the start of the event. Because of the configuration, that is the plant was in a refueling outage, this was unavoidable.
- Defenses against human errors are preserved: This was not maintained.
- The intent of the plant's design criteria is maintained: Indeterminant.

From the review above it is clear that many of the defense in depth criteria capabilities were not strong during the event. The other aspect of a proper deterministic analysis includes a safety margin review. Based on the RG 1.174 definition this event does not appear to challenge the plant's applicable safety margin.

(Pages A-26 to A-27)

Question 4: Containment isolated or not bypassed?

Dry well head is removed in preparation for refueling; therefore, containment is open.

Question 7: Core damage arrest without vessel breach (VB)?

The reactor head vent piping is removed thus the vessel is breached in this scenario.

Question 8: Containment failure at vessel breach (VB)?

This question does not need to be evaluated based on the applicable paths through the event tree. However, the dry well head which is the containment head is removed in preparation of refueling therefore, containment is failed.

… Question 10: No potential for early fatalities?

… As the dry well (containment) head is removed, this is evaluated as a containment failure. Thus radionuclide release from containment happens at the same time as core damage. The time of declaring an emergency in relationship to the predicted core damage of nine hours. The assumption of when the emergency response organization would order an evaluation of the population closest to the plant becomes a major factor is determining what LERF factor values to use. A protective action recommendation is not made by this site under Genera l Emergency; however, the state/county may order evacuations when the Site Area Emergency level is reached.